INL/EXT-18-51911-Revision-0

ECAR-4310, Evaluation of the Inhalation Dose Consequences for the HALEU Environmental Assessment

Boyd D Christensen

October 2018

The INL is a U.S. Department of Energy National Laboratory
operated by Battelle Energy Alliance

INL/EXT-18-51911-Revision-0

ECAR-4310, Evaluation of the Inhalation Dose Consequences for the HALEU Environmental Assessment

Boyd D Christensen

October 2018

Idaho National Laboratory
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Environment, Safety and Health
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

ENGINEERING CALCULATIONS AND ANALYSIS

Title: Evaluation of the Inhalation Dose Consequences for the HALEU Environmental Assessment

ECAR No.: 4310 Rev. No.: 0 Project No.: N/A Date: 10/30/18

1. Does this ECAR involve a Safety SSC?	N/A	**Professional Engineer's Stamp**
2. Safety SSC Determination Document ID	N/A	PE Stamp Not Required per LWP-10010.
3. Engineering Job (EJ) No.	N/A	
4. SSC ID	N/A	
5. Building	TBD	
6. Site Area	MFC/INTEC	

7. Objective/Purpose:

The U. S. Department of Energy (DOE) proposes to expand the fuel fabrication capability at Idaho National Laboratory (INL) to produce needed quantities of High-Assay Low-Enriched Uranium (HALEU) fuel from down-blended EBR-II fuel processed at the Fuels Conditioning Facility (FCF) and stored at INL. The expansion of the fuel fabrication capability would include the purchase of new equipment and use of existing facilities at INL's Materials Fuels Complex (MFC) and the Idaho Nuclear Technology and Engineering Center (INTEC).

The purpose of this report is to document calculation of the estimated potential inhalation dose to members of the public and workers from accidental air emissions associated with production of HALEU fuel. This document supports the Environmental Assessment for Use of DOE-Owned HALEU Stored at Idaho National Laboratory.

8. If revision, please state the reason and list sections and/or pages being affected: N/A

Title: Evaluation of the Inhalation Dose Consequences for the HALEU Environmental Assessment

ECAR No.: 4310 Rev. No.: 0 Project No.: N/A Date: 10/30/18

9. Conclusions/Recommendations:

Potential inhalation dose to members of the public and workers are calculated for receptors at INTEC and MFC and at the closest site boundary. The analyzed event is a spill or free-fall drop of molten process material during heating. The inhalation dose consequences are summarized below.

Receptor	Source	Dose[1]
Facility Worker	CED (rem/minute)	8.81E+00
Collocated Worker (100 m)	CED	9.97E-01
	cloud gamma	8.98E-07
	total	9.97E-01
Public (5,000 m)	CED	2.94E-02
	cloud gamma	1.15E-08
	total	2.94E-02
Public (14,000 m)	CED	9.10E-03
	cloud gamma	3.57E-9
	total	9.10E-03

1. All values are in rem, except Facility Worker CED is dose rate in rem for 1-minute exposure.

The dose consequences reported in this table are considered bounding given the conservative assumptions of the radioactive material release. The accident was analyzed as unmitigated with no credit taken for engineering or administrative controls or safety systems. These results provide a technical basis for evaluating potential consequences from inhalation doses associated with postulated accidents from HALEU fuel fabrication at INL in support of the EA.

CONTENTS

PROJECT ROLES AND RESPONSIBILITIES 4

1 SCOPE AND BRIEF DESCRIPTION 5

2 DESIGN OR TECHNICAL PARAMETER INPUT AND SOURCES 5

3 RESULTS OF LITERATURE SEARCHES AND OTHER BACKGROUND DATA .. 5

4 ASSUMPTIONS 5

5 COMPUTER CODE VALIDATION 6

6 DISCUSSION/ANALYSIS 6

6.1 Source Term 7

6.2 Radionuclide Distributions for HALEU Radioactive Material Inventories 7

6.3 Release Parameters 8

6.3.1 Material-at-Risk 9

6.3.2 Damage Ratio 9

6.3.3 Airborne Release Fraction 9

6.3.4 Respirable Fraction 9

6.3.5 Leak Path Factor 9

6.4 Committed Effective Dose 10

6.5 Downwind Exposures 11

6.6 Bounding Inhalation Dose Consequences 12

7 REFERENCES 13

APPENDIX A, RSAC Runs

Title: Evaluation of the Inhalation Dose Consequences for the HALEU Environmental Assessment

ECAR No.: 4310 Rev. No.: 0 Project No.: N/A Date: 10/30/18

PROJECT ROLES AND RESPONSIBILITIES

Project Role	Name (Typed)	Organization	Pages covered (if applicable)
Performer	Boyd Christensen	H372	See eCR 663656.
Checker[a]	Noel Duckwitz	H372	See eCR 663656.
Independent Reviewer[b]	Tim Solle	H510	See eCR 663656.
CUI Reviewer[c]	N/A	N/A	N/A
Manager[d]	Brady Orchard	F500	See eCR 663656.
Requestor[e]	John Irving	H520	See eCR 663656.
Nuclear Safety[e]	Noel Duckwitz	H372	See eCR 663656.
Document Owner[e]	Brady Orchard	F500	See eCR 663656.

Responsibilities:

a. Confirmation of completeness, mathematical accuracy, and correctness of data and appropriateness of assumptions.

b. Concurrence of method or approach. See definition, LWP-10106.

c. Concurrence with the document's markings in accordance with LWP-11202.

d. Concurrence of procedure compliance. Concurrence with method/approach and conclusion.

e. Concurrence with the document's assumptions and input information. See definition of Acceptance, LWP-10200.

NOTE: *Delete or mark "N/A" for project roles not engaged. Include ALL personnel and their roles listed above in the eCR system. The list of the roles above is not all-inclusive. If needed, the list can be extended or reduced.*

1 SCOPE AND BRIEF DESCRIPTION

The U. S. Department of Energy (DOE) proposes to expand the fuel fabrication capability at Idaho National Laboratory (INL) to produce needed quantities of High-Assay Low-Enriched Uranium (HALEU) fuel from HALEU material stored at INL. The expansion of the fuel fabrication capability would include the purchase of new equipment and use of existing facilities at INL's Materials Fuels Complex (MFC) and the Idaho Nuclear Technology and Engineering Center (INTEC). The purpose of this report is to document calculation of the estimated potential dose to members of the public and workers from accidental air emissions associated with production of HALEU fuel. This document supports the *Environmental Assessment for Use of DOE-Owned HALEU Stored at Idaho National Laboratory*[1].

The Environmental Assessment (EA) proposes that up to 5,000 kg of HALEU material be processed annually at INL. Alternative 1a of the EA proposes that up to 2,500 kg be processed at each of two facilities at MFC, and Alternative 1b proposes that up to 2,500 kg be processed at an MFC facility and 2,500 kg processed at an INTEC facility.

Dose calculations are made for public receptors, collocated workers, and facility workers from a bounding postulated accident in either INTEC or MFC facilities.

2 DESIGN OR TECHNICAL PARAMETER INPUT AND SOURCES

Technical parameter input and sources are identified in the text as appropriate.

3 RESULTS OF LITERATURE SEARCHES AND OTHER BACKGROUND DATA

The sources used to develop radionuclide inventories and bounding dose consequences are identified in the text as appropriate.

4 ASSUMPTIONS

The following assumptions were used in evaluating dose consequence to workers and the public:

- Batch processing is up to 50 kg HALEU feedstock material
- Two batches may be in process simultaneously
- HALEU feedstock material is stored in closed containers when not in process and therefore not considered at risk in this analysis
- Unlike plutonium, uranium is difficult to ignite (DOE-HDBK-3010-94) especially with large pieces such as ingots
- Performance of the HALEU material in accidents (thermal stress, spill, drop, impact) follow that of uranium. The HALEU is >99% uranium. The other elements identified are homogeneously entrained throughout the uranium matrix on ppm and ppb scale
- In the event of an accident involving radioactive material, it is assumed that the facility worker(s) would evacuate the affected facility in less than 1 minute and relocate to an unaffected area per facility worker training
- This accident was analyzed as unmitigated with no credit taken for engineered controls or safety systems such as glovebox or ventilation systems
- Molten HALEU material spills occur at a height of <4 m. Process equipment would be placed on the floor of the work area, and material would be molten only during processing

- A room/building volume of 1,700 m^3 was assumed for facility worker dose calculations. For smaller or larger areas, higher or lower dose consequences can be scaled from results documented herein
- The nearest public receptor is located at the nearest INL site boundary (5,000 m from MFC and 14,000 m from INTEC)
- The composition of HALEU feedstock material is based upon the best available information and values are shown within this document.

5 COMPUTER CODE VALIDATION

A. Computer type:

Dell Optiplex 7010

Property Tag Number: 426810

B. Operating System and Version: Windows 7 Enterprise

C. Computer program name and revision: Radiological Safety Analysis Computer Program (RSAC) 7.2. RSAC 7.2 program is a radiological safety analysis program that has been used extensively at the INL for calculating the doses to facility worker, collocated workers, and off-site public due to radiological releases. It has been independently verified and validated for these types of calculations. Evidence of, or reference to, computer program validation: RSAC 7.2 configuration management is maintained in Enterprise Architecture under configuration number 223954. RSAC 7.2 was used to calculate final dose consequences for this ECAR. Case study verification for program installation on this computer platform was successfully completed by running each of the 20 example problems under the RSAC QC menu option.

6 DISCUSSION/ANALYSIS

The needs of the proposed project to fabricate oxide and metallic HALEU fuel at INL facilities are anticipated to be filled using more than one facility on the INL site. The project will use either multiple facilities at MFC, or facilities at both MFC and INTEC.

Dose consequence to the facility worker in an accident varies depending on the size of the room or building in which the accident occurs. Dose consequence to the collocated worker is calculated at 100 m and is not dependent on location. Dose consequence to the public is dependent upon the location of the activity and corresponding distance to the nearest INL site boundary. Accident consequences in this analysis will focus on molten material spill occurring at both the INTEC complex and MFC complex and will use nominal values for building size and distance to the public.

Accidents considered for HALEU processing include thermal stress of fire, spill or free-fall drop of molten metal, and accidents resulting in solid ingot free-fall drop or impact. Due to the amount of HALEU potentially present in process facilities, the potential for accidental nuclear criticality exists. Should DOE proceed with this project to fabricate HALEU fuel, and once specific siting locations are determined, criticality hazards will be evaluated and documented in a separate analysis in accordance with DOE-STD-3007-2007, Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Non-Reactor Nuclear Facilities, and in support of facility safety basis development.

Of the postulated accidents, material at risk (MAR) is the same in each case. It is anticipated that batches in process will not exceed 50 kg. In order to achieve the production requirements of the program, it is possible that two process lines could operate simultaneously and so for conservative analysis, double batch upsets are analyzed under accident conditions for airborne release. Release fractions may be different for the various accidents under consideration. DOE-HDBK-3010-94, “Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities,”[2] states that no significant airborne release is postulated from spill and impact of solid uranium, so those types of events are not addressed further in this document. The handbook also states that bounding airborne release fraction (ARF) and respirable fraction (RF) values combined for uranium pieces or powder under thermal stress (fire) is ARF x RF = 1.0E-3, and for disturbed molten metal surface under dynamic conditions such as a spill or free-fall drop (from 2 to 4 m) is ARF x RF = 6.0E-3. Therefore, this analysis will focus on spill or drop of molten uranium during HALEU material processing.

6.1 Source Term

The source term is represented by a portion of the HALEU feedstock material consisting of two batches of 50 kg each, of homogeneous uranium product with impurities of other isotopes as identified below. The source term for accident analysis was evaluated based on calculated and measured analytical results which are documented and summarized in TEV-3537, “Isotopic Characterization of HALEU from EBR-II Driver Fuel Processing.”[3] This has been identified as a reasonable estimate of material that bounds the expected activity.

6.2 Radionuclide Distributions for HALEU Radioactive Material Inventories

The following table shows the expected radionuclide composition for the assumed 100 kg of HALEU, which may be in process at any given time. The table is based on maximum hypothetical values using the highest evaluated isotopic concentration of each isotope in HALEU feedstock material samples from TEV-3537, and produces a bounding radioactivity (Ci) of each radioisotope expected to be present.

Conversion to Curie									
Radionuclide	Composition (TEV-3537)[a]	Units (TEV-3537)	Total Weight U (g)	Total Weight of each U Radionuclide (g)	Conversion to g/kg	Mass (kg)	Isotope Mass (g)	Sp. Activity (Ci/g)	Activity (Ci)
Mn-54	1.25E-05	ppm (conv from ppt)			1.25E-08	100	1.25E-06	7.75E+03	9.68E-03
Co-60	1.77E-04	ppm (conv from ppt)			1.77E-07	100	1.77E-05	1.13E+03	2.00E-02
Sr-90	6.25E-02	ppm (conv from ppb)			6.25E-05	100	6.25E-03	1.38E+02	8.63E-01
Tc-99	2.80E-01	ppm			2.80E-04	100	2.80E-02	1.70E-02	4.75E-04
Sb-125	4.81E-04	ppm (conv from ppt)			4.81E-07	100	4.81E-05	1.04E+03	4.99E-02
Cs-134	1.54E-04	ppm (conv from ppt)			1.54E-07	100	1.54E-05	1.29E+03	1.99E-02
Cs-135	3.00E+00	ppm			3.00E-03	100	3.00E-01	1.12E-03	3.37E-04
Cs-137	2.24E-02	ppm (conv from ppb)			2.24E-05	100	2.24E-03	8.68E+01	1.94E-01
Ce-144	3.13E-04	ppm (conv from ppt)			3.13E-07	100	3.13E-05	3.19E+03	9.97E-02
Eu-154	1.11E-03	ppm (conv from ppb)			1.11E-06	100	1.11E-04	2.70E+02	3.00E-02
Eu-155	1.04E-03	ppm (conv from ppb)			1.04E-06	100	1.04E-04	4.86E+02	5.05E-02
Np-237	2.14E+01	ppm			2.14E-02	100	2.14E+00	7.05E-04	1.51E-03
Pu-239	1.03E+02	ppm			1.03E-01	100	1.03E+01	6.21E-02	6.40E-01
Pu-240	2.71E+00	ppm			2.71E-03	100	2.71E-01	2.27E-01	6.16E-02
Am-241	2.92E-01	ppm (conv from ppb)			2.92E-04	100	2.92E-02	3.43E+00	1.00E-01
U-234	0.18%	wt % of U	99990	1.80E+02			1.80E+02	6.21E-03	1.12E+00
U-235	19.42%	wt % of U	99990	1.94E+04			1.94E+04	2.16E-06	4.20E-02
U-236	0.56%	wt % of U	99990	5.60E+02			5.60E+02	6.47E-05	3.62E-02
U-238	80.03%	wt % of U	99990	8.00E+04			8.00E+04	3.36E-07	2.69E-02
U-232	5.04E-03	ppmU (conv from ppbU)			5.04E-06	99.97	5.04E-04	2.21E+01	1.11E-02
U-233	3.18E-01	ppmU (conv from ppbU)			3.18E-04	99.97	3.18E-02	9.64E-03	3.07E-04
U-237	2.20E-07	ppmU (conv from pptU)			2.20E-10	99.97	2.20E-08	8.16E+04	1.79E-03
Max measured values from TEV-3537 Appendix C			0.9999 wt%U from TEV-3537 App C			0.9997wt%U from TEV-3537 App B			
Max calc'd ppbU/pptU from TEV-3537 Appendix B									
[a] TEV-3537, *Isotopic Characterization of HALEU from EBR-II Driver Fuel Processing*, Rev 0, October 2018									

Table 1. Expected radionuclide composition.

6.3 Release Parameters

The accident-specific parameters used to evaluate the dose to downwind receptors requires that certain assumptions be made that modify the dispersion release fraction due to the physical aspects of the release. The five components of the following source-term equation recommended by DOE-HDBK-3010-94 contain the basis for the material release parameters. To calculate downwind radiological doses for these scenarios, a source term (ST) was determined. The ST is the amount of radioactive material released during the postulated accident scenario. The ST is determined using the following equation:

$$ST = MAR \times DR \times ARF \times RF \times LPF$$

where

ST = source term

MAR = material-at-risk

DR	=	damage ratio
ARF	=	airborne release fraction
RF	=	respirable fraction
LPF	=	leak path factor

6.3.1 Material-at-Risk

The material-at-risk (MAR) is the total inventory that could be impacted for a given accident scenario and is expressed in terms of total quantity at risk. The radiological inventory used for this accident analysis is based on the 100 kg of HALEU feedstock material that may be stored in a given facility at any given time. This MAR is comprised of the radionuclide distribution as shown in the table in Section 6.2 above.

6.3.2 Damage Ratio

The damage ratio (DR) represents the fraction of MAR that could be affected by the postulated accident and is a function of the accident initiator and the operational scenario being evaluated. DRs are determined based on engineering judgment, best available information, and prior analyses. The DR for the scenario being evaluated is 1. It assumes that there are two 50 kg batches being processed simultaneously at the time of the initiating event and the material from those two batches are subject to release mechanisms.

6.3.3 Airborne Release Fraction

The ARF is the coefficient used to estimate the amount of material suspended in the air as an aerosol, thus becoming available for transport. The ARF is related to the physical stresses of a specific accident and the physical characteristics of the material involved in the accident. DOE-HDBK-3010-94 is used to derive the ARF. The HALEU MAR is conservatively modeled as a molten liquid under dynamic conditions such as a spill or free-fall drop, and the ARF is 6.0E-3 with a fall distance between 2 and 4 m.

6.3.4 Respirable Fraction

The RF is the fraction of airborne particles that can be transported through air and inhaled into the pulmonary region of the human respiratory system. The RF includes particles having a 10-μm aerodynamic equivalent diameter or less. DOE-HDBK-3010-94 is used to derive the RF. The HALEU MAR is modeled as a molten liquid, and the RF for this analysis conservatively assumed to be 1.0.

6.3.5 Leak Path Factor

The leak path factor (LPF) is the fraction of material in the aerosol transported through some confinement deposition or filtration mechanism. Typically, this factor accounts for the amount of the airborne MAR that escapes from a building or room. LPFs are assumed to be 1.0 to ensure a conservative and unmitigated analysis.

6.3.6 Source Term

Using the source term equation above, the following table derives the source term for each of the radioisotopes using the five factors for release.

Radionuclide	Activity (Ci)	DR	ARF	RF	LPF	ST
Mn-54	9.68E-03	1	6.00E-03	1	1	5.81E-05
Co-60	2.00E-02	1	6.00E-03	1	1	1.20E-04
Sr-90	8.63E-01	1	6.00E-03	1	1	5.18E-03
Tc-99	4.75E-04	1	6.00E-03	1	1	2.85E-06
Sb-125	4.99E-02	1	6.00E-03	1	1	2.99E-04
Cs-134	1.99E-02	1	6.00E-03	1	1	1.19E-04
Cs-135	3.37E-04	1	6.00E-03	1	1	2.02E-06
Cs-137	1.94E-01	1	6.00E-03	1	1	1.17E-03
Ce-144	9.97E-02	1	6.00E-03	1	1	5.98E-04
Eu-154	3.00E-02	1	6.00E-03	1	1	1.80E-04
Eu-155	5.05E-02	1	6.00E-03	1	1	3.03E-04
Np-237	1.51E-03	1	6.00E-03	1	1	9.05E-06
Pu-239	6.40E-01	1	6.00E-03	1	1	3.84E-03
Pu-240	6.16E-02	1	6.00E-03	1	1	3.69E-04
Am-241	1.00E-01	1	6.00E-03	1	1	6.01E-04
U-234	1.12E+00	1	6.00E-03	1	1	6.71E-03
U-235	4.20E-02	1	6.00E-03	1	1	2.52E-04
U-236	3.62E-02	1	6.00E-03	1	1	2.17E-04
U-238	2.69E-02	1	6.00E-03	1	1	1.61E-04
U-232	1.11E-02	1	6.00E-03	1	1	6.68E-05
U-233	3.07E-04	1	6.00E-03	1	1	1.84E-06
U-237	1.79E-03	1	6.00E-03	1	1	1.08E-05

Table 2. Radionuclide source term.

6.4 Committed Effective Dose

The committed effective dose (CED) for downwind receptors is estimated from:

$$CED = ST \times \chi/Q \times BR \times DCF \times DDF$$

Where

CED = committed effective dose

ST	=	source term (Ci)
χ/Q	=	plume dispersion (s/m^3)
BR	=	breathing rate (m/s^3)
DCF	=	dose conversion factor (rem/ci)
DDF	=	fraction of radionuclide remaining in plume after dry deposition (no units)

The χ/Q plume dispersion value is a function of the meteorological conditions involved in the accident and relative location of the release point and the receptor and is calculated in RSAC. χ/Q is considered to be constant for all radionuclides of respirable size in an event and is independent of particular radionuclide and source types. Breathing rate (BR) is the assumed breathing rate described in DOE O 440.1B, "Worker Protection Program for DOE,"[4] and is 3.33E 4 m^3/sec. ICRP-68, "Dose Coefficients for Intakes of Radionuclides by Workers,"[5] DCF values were utilized in the CED calculation for facility and collocated workers, and ICRP 72, "Age-dependent Doses from Intakes of Radionuclides,"[6] DCF values were utilized in the calculation of the CED for the public. The dry deposition factor (DDF) accounts for the material that is removed from the plume via deposition on the ground and accounts only for plume depletion that is expected to occur, independent of weather conditions.

For facility worker, the model calculates dose based on release to a room with a nominal volume of 1,700 m^3 based on a moderate size building or room. Smaller or larger buildings or rooms will result in proportionately higher or lower dose estimates to the facility worker from what is estimated here.

6.5 Downwind Exposures

RSAC-7.2 was used to quantify the doses of the postulated accident. The program is used to calculate the doses from the release of radionuclides to the atmosphere. The meteorological capabilities of RSAC 7.2 include Gaussian plume diffusion for the Pasquill Gifford, Hilsmeier-Gifford, and Markee diffusion models. The Markee model is used in this analysis for all downwind receptors because it was developed specifically for INL sagebrush terrain for effluent releases from a few minutes to 15 minutes in duration. The RSAC input parameters are summarized below.

RSAC parameters downwind scenario.

RSAC Input Parameters	Input Values
Release elevation (m)	0
Stability class	F
Wind speed (m/second)	1.04
Diffusion coefficient	Markee
Downwind receptor distance (m)	100, 5000, 14000
Breathing rate (m^3/second)	3.33E-04

The RSAC-7 program allows the user to specify meteorological conditions at the time of radiological release and to calculate diffusion, dispersion, and depletion factors. It also allows the user to perform a variety of dose calculations. This analysis reports estimated dose to the whole body of the receptor and does not identify specific organ doses. Doses from the ingestion, ground contamination, and air immersion exposure pathways are negligible so for this analysis, CED = TED (total effective dose).

Univariate distributions of wind conditions were developed by combining wind speed class and stability class into one variable (i.e., wind condition) and deriving the probability distribution of this variable. Cumulative probability distributions were then developed based on rankings of the wind conditions, ranging from good to bad in terms of their effect on dispersion of airborne contaminants. Based on this work and 95% meteorology conditions, a stability class of F and a wind speed class of 1 (i.e., 1.04 m/second) were used for the RSAC-7.2 runs. A ground release was assumed for this scenario. Buoyant plume rise was not used. Receptor locations were at 100 m downwind from the release for the collocated worker, and a distances of 5,000 m and 14,000 m were used for determining the off-site public dose, which is the nearest site boundary to MFC and INTEC respectively.

6.6 Bounding Inhalation Dose Consequences

The calculation of 100 kg molten HALEU feedstock material was postulated to be involved in a spill under environmental conditions as described above. The results from the RSAC consequence calculations are summarized in the table below.

Receptor	Source	Dose[1]
Facility Worker	CED (rem/minute)	8.81E+00
Collocated Worker (100 m)	CED	9.97E-01
	cloud gamma	8.98E-07
	total	9.97E-01
Public (5,000 m)	CED	2.94E-02
	cloud gamma	1.15E-08
	total	2.94E-02
Public (14,000 m)	CED	9.10E-03
	cloud gamma	3.57E-9
	total	9.10E-03

1. All values are in rem, except Facility Worker CED is dose rate in rem for 1-minute exposure.

Table 3. Inhalation dose consequences for HALEU.

The dose consequences reported in this table are considered bounding given the conservative assumptions of the radioactive material release. The accident was analyzed as unmitigated with no credit taken for engineering or administrative controls or safety systems. These results provide a technical basis for evaluating potential consequences from inhalation doses associated with postulated accidents from HALEU fuel fabrication at INL in support of the EA.

7 REFERENCES

1. DOE/EA-2087, Environmental Assessment for Use of DOE-Owned High-Assay Low-Enriched Uranium Stored at Idaho National Laboratory, U.S. Department of Energy, 2018.

2. DOE-HDBK-3010-94, "Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities," Change 1, U.S. Department of Energy, March 2000.

3. TEV-3537, "Isotopic Characterization of HALEU from EBR-II Driver Fuel Processing," Rev 0, October 2018.

4. DOE O 440.1B, "Worker Protection Program for DOE," U.S. Department of Energy, May 2007.

5. ICRP-68, "Dose Coefficients for Intakes of Radionuclides by Workers," International Commission on Radiological Protection, 1994.

6. ICRP 72, "Age-dependent Doses from Intakes of Radionuclides," International Commission on Radiological Protection, 1996.

APPENDIX A

RSAC Runs

Table A-1. RSAC output file for HALEU Facility Worker dose.

Radiological Safety Analysis Computer Program (RSAC 7.2.0)

Name: INL **Company:** Idaho National Laboratory **Serial:** OISSE-77YKP-GW43J

Computer: INL426810 **Run Date:** 10/25/2018 **Run Time:** 14:28:55

File: HALEU 100 kg batch molten 1700 m^3.rsac

Input

```
* Double-click to specify a title
# Double-click to add a description for this file
#
#
# 100 kg HALEU batch
2000,1,0
# Original file copied to temporary external file
2002,TempSrc1.txt
# 2002,C:\Users\chrib\Documents\Rsac\Input Files\HALEU 100 kg inventory.txt
2999
# Inside 1700 m^3 bldg
5000,1
5500,1700.
5999
# Affected Worker Dose
7000,1,0,1,0,2,7
7001,3.33E-04,0.,0,0,1.
7002,26
7999
# cloud gamma
9000,0,0.
10000
```

Direct Radionuclide Input

```
       PREVIOUS INVENTORY CHANGED TO THE FOLLOWING VALUES
     RADIONUCLIDE INPUT READ FROM EXTERNAL FILE USER FILE   TempSrc1.txt
     NUCLIDE            HALF LIFE       CURIE
  932370   Np237    2.144E+06  yr    9.050E-06
  922380    U238    4.468E+09  yr    1.610E-04
  922360    U236    2.342E+07  yr    2.170E-04
  922350    U235    7.038E+08  yr    2.520E-04
  942400   Pu240    6.561E+03  yr    3.690E-04
  551370   Cs137    3.008E+01  yr    1.170E-03
  380900   Sr 90    2.879E+01  yr    5.180E-03
  942390   Pu239    2.411E+04  yr    3.840E-03
  922340    U234    2.455E+05  yr    6.710E-03
  250540   Mn 54    3.120E+02  d     5.810E-05
```

```
  270600  Co 60    5.275E+00  yr    1.200E-04
  430990  Tc 99    2.111E+05  yr    2.850E-06
  511250  Sb125    2.759E+00  yr    2.990E-04
  551340  Cs134    2.065E+00  yr    1.190E-04
  551350  Cs135    2.300E+06  yr    2.020E-06
  581440  Ce144    2.849E+02  d     5.980E-04
  631540  Eu154    8.601E+00  yr    1.800E-04
  631550  Eu155    4.753E+00  yr    3.030E-04
  952410  Am241    4.326E+02  yr    6.010E-04
  922320   U232    6.890E+01  yr    6.680E-05
  922330   U233    1.592E+05  yr    1.840E-06
  922370   U237    6.750E+00  d     1.080E-05
***ROOM DISPERSION DATA
    BUILDING ROOM VOLUME =  1.700E+03 cu m
```

Inhalation Dose Calculation

```
         USING DOSE CONVERSION FACTORS FROM ICRP-68 FOR ADULT WORKER
    RESPIRABLE FRACTION =  1.000E+00
    BREATHING RATE =  3.330E-04 (m^3/s)
    TIME RECEPTOR REMAINS IN BUILDING ROOM =  6.000E+01 (s)
    INTERNAL EXPOSURE TIME PERIOD  =  5.000E+01 (yr)
    LUNG ABSORPTION TYPES SELECTED TO GIVE MAXIMUM DOSE
    ICRP-68 INHALATION DOSE FOR ADULT WORKER
           E_50 DOSE                   EQUIVALENT   EFFECTIVE
                        TOTAL Mn                       3.03E-06
                        TOTAL Co                       8.87E-05
                        TOTAL Sr                       1.73E-02
                        TOTAL  Y                       3.46E-08
                        TOTAL Tc                       3.97E-07
                        TOTAL Sb                       4.29E-05
                        TOTAL Te                       1.68E-12
                        TOTAL Cs                       3.91E-04
                        TOTAL Ce                       7.54E-04
                        TOTAL Pr                       1.54E-08
                        TOTAL Eu                       3.36E-04
                        TOTAL Pb                       3.23E-08
                        TOTAL Bi                       2.37E-09
                        TOTAL Po                       6.44E-08
                        TOTAL Ra                       6.61E-08
                        TOTAL Ac                       8.06E-11
                        TOTAL Th                       2.14E-07
                        TOTAL Pa                       4.89E-09
                        TOTAL  U                       2.23E+00
                        TOTAL Np                       5.90E-03
                        TOTAL Pu                       5.86E+00
                        TOTAL Am                       7.06E-01
                TOTAL     E_50 DOSE =                  8.81E+00
ONLY INHALATION DOSE CAN BE CALCULATED FOR RECEPTOR IN ROOM
```

Execution Time

```
           1.00E-02 SECONDS
```

Table A-2. RSAC output file for HALEU Collocated Worker dose rate (100 m).

Radiological Safety Analysis Computer Program (RSAC 7.2.0)

Name: INL **Company:** Idaho National Laboratory **Serial:** OISSE-77YKP-GW43J

Computer: INL426810 **Run Date:** 10/25/2018 **Run Time:** 14:25:30

File: HALEU 100 kg batch molten spill at 100m.rsac

Input

```
* Double-click to specify a title
# Double-click to add a description for this file
#
#
# 100 kg HALEU batch. Molten Spill. ARFxRF = 6.0E-3
2000,1,0
# Original file copied to temporary external file
2002,TempSrc1.txt
# 2002,C:\Users\chrib\Documents\Rsac\Input Files\HALEU 100 kg inventory.txt
2999
# Collocated worker at 100m
5000,0
5001,1.04,0.,400.,1.099E3,0.,1
5002,0.001,0.01,0.,0.001,0.001
5101,100.
5201,1.,0.
5400,2,0.,0.
5410,2,6,0,0.
5999
# cloud gamma
9000,1,0.
# Collocated Worker Dose Calc at 100 m
7000,1,0,1,0,2,7
7001,3.33E-04,0.,0,0,1.
7002,26
7999
10000
```

Direct Radionuclide Input

```
        PREVIOUS INVENTORY CHANGED TO THE FOLLOWING VALUES
     RADIONUCLIDE INPUT READ FROM EXTERNAL FILE USER FILE   TempSrc1.txt
    NUCLIDE           HALF LIFE       CURIE
  932370   Np237    2.144E+06   yr    9.050E-06
  922380    U238    4.468E+09   yr    1.610E-04
  922360    U236    2.342E+07   yr    2.170E-04
  922350    U235    7.038E+08   yr    2.520E-04
  942400   Pu240    6.561E+03   yr    3.690E-04
  551370   Cs137    3.008E+01   yr    1.170E-03
  380900   Sr 90    2.879E+01   yr    5.180E-03
  942390   Pu239    2.411E+04   yr    3.840E-03
  922340    U234    2.455E+05   yr    6.710E-03
```

```
  250540  Mn 54    3.120E+02  d     5.810E-05
  270600  Co 60    5.275E+00  yr    1.200E-04
  430990  Tc 99    2.111E+05  yr    2.850E-06
  511250  Sb125    2.759E+00  yr    2.990E-04
  551340  Cs134    2.065E+00  yr    1.190E-04
  551350  Cs135    2.300E+06  yr    2.020E-06
  581440  Ce144    2.849E+02  d     5.980E-04
  631540  Eu154    8.601E+00  yr    1.800E-04
  631550  Eu155    4.753E+00  yr    3.030E-04
  952410  Am241    4.326E+02  yr    6.010E-04
  922320   U232    6.890E+01  yr    6.680E-05
  922330   U233    1.592E+05  yr    1.840E-06
  922370   U237    6.750E+00  d     1.080E-05
```

Meteorological Data

```
     MEAN WIND SPEED = 1.040E+00 (m/s)   STACK HEIGHT = 0.000E+00 (m)
 MIXING LAYER HEIGHT = 4.000E+02 (m)      AIR DENSITY = 1.099E+03 (g/cu m)
 WET DEPOSITION SCAVENGING COEFFICIENT = 0.000E+00 (1/s)
 DRY DEPOSITION VELOCITIES (m/s)
      SOLIDS = 1.000E-03    HALOGENS = 1.000E-02   NOBLE GASES = 0.000E+00
      CESIUM = 1.000E-03   RUTHENIUM = 1.000E-03
 THERE IS 1 SET OF LEAKAGE CONSTANTS (K1,K2)
       1.000E+00       0.000E+00
 PLUME MEANDER FACTOR = 1.00E+00
 PASQUILL CLASS F METEOROLOGY, MARKEE SIGMA VALUES
 NO BUILDING WAKE CORRECTION MADE
   DOWNWIND DISTANCE     STACK         SIGY        SIGZ        CHI/Q
                        HEIGHT (m)     (m)         (m)        (s/m^3)
      1.000E+02        0.000E+00   2.069E+01   3.625E+00   4.081E-03
 PLUME DEPLETION BY FALLOUT IS INCLUDED
   FRACTION OF PLUME REMAINING AIRBORNE FOLLOWING DEPLETION BY DEPOSITION
     DOWNWIND DISTANCE     SOLIDS   HALOGENS    CESIUM    RUTHENIUM
         1.000E+02       9.780E-01  8.008E-01  9.780E-01  9.780E-01
```

Gamma Dose Calculation

```
    EXPOSURE TIME =  1.0000E+00 (S)
   CALCULATIONS MADE USING THE SEMI-INFINITE MODEL
                                                EXTERNAL EDE
  DOWNWIND DISTANCE =  1.000E+02 (M)             DOSE =  8.98E-07 (REM)
```

Inhalation Dose Calculation

```
      USING DOSE CONVERSION FACTORS FROM ICRP-68 FOR ADULT WORKER
 RESPIRABLE FRACTION =  1.000E+00
 BREATHING RATE =  3.330E-04 (m^3/s)
 RELEASE TIME FOR EXPONENTIAL DECAY FUNCTION =  1.000E+00 (s)
 INTERNAL EXPOSURE TIME PERIOD  =  5.000E+01 (yr)
```

```
    LUNG ABSORPTION TYPES SELECTED TO GIVE MAXIMUM DOSE
    ICRP-68 INHALATION DOSE FOR ADULT WORKER
    ICRP-68 INHALATION DOSE (rem)          CHI/Q =  4.081E-03 (s/m^3)
      DOWNWIND DISTANCE = 1.00E+02 (m)     PLUME TRAVEL TIME = 9.62E+01 (s)
           E_50 DOSE                    EQUIVALENT   EFFECTIVE
     ICRP-72 INHALATION DOSE      DOWNWIND DISTANCE = 1.000E+02 m
    ICRP-68 INHALATION DOSE                CHI/Q =  4.081E-03 (s/cu m)
      DOWNWIND DISTANCE = 1.00E+02 (m)     PLUME TRAVEL TIME = 9.62E+01 (s)
                         TOTAL Mn                    3.43E-07
                         TOTAL Co                    1.00E-05
                         TOTAL Sr                    1.96E-03
                         TOTAL  Y                    1.25E-08
                         TOTAL Tc                    4.48E-08
                         TOTAL Sb                    4.85E-06
                         TOTAL Te                    5.73E-11
                         TOTAL Cs                    4.42E-05
                         TOTAL Ce                    8.53E-05
                         TOTAL Pr                    5.52E-09
                         TOTAL Eu                    3.80E-05
                         TOTAL Pb                    2.90E-19
                         TOTAL Ra                    7.01E-16
                         TOTAL Ac                    1.19E-13
                         TOTAL Th                    9.58E-09
                         TOTAL Pa                    4.07E-12
                         TOTAL  U                    2.52E-01
                         TOTAL Np                    6.68E-04
                         TOTAL Pu                    6.62E-01
                         TOTAL Am                    7.98E-02
                TOTAL     E_50 DOSE =                9.97E-01
```

Execution Time

```
           1.00E-02 SECONDS
```

Table A-3. RSAC output file for HALEU Public dose (5,000 m)

Radiological Safety Analysis Computer Program (RSAC 7.2.0)

Name: INL **Company:** Idaho National Laboratory **Serial:** OISSE-77YKP-GW43J

Computer: INL426810 **Run Date:** 10/25/2018 **Run Time:** 14:26:10

File: HALEU 100 kg batch molten spill at 5000m.rsac

Input

```
* Double-click to specify a title
# Double-click to add a description for this file
#
#
# 100 kg HALEU batch. Molten spill. ARFxRF = 6.0E-3
2000,1,0
# Original file copied to temporary external file
2002,TempSrc1.txt
# 2002,C:\Users\chrib\Documents\Rsac\Input Files\HALEU 100 kg inventory.txt
2999
# Public at 5000m
5000,0
5001,1.04,0.,400.,1.099E3,0.,1
5002,0.001,0.01,0.,0.001,0.001
5101,5000.
5201,1.,0.
5400,2,0.,0.
5410,2,6,0,0.
5999
# Public Dose
7000,1,0,1,0,2,6
7001,3.33E-04,0.,0,0,1.
7002,26
7999
# cloud gamma
9000,1,0.
10000
```

Direct Radionuclide Input

```
        PREVIOUS INVENTORY CHANGED TO THE FOLLOWING VALUES
     RADIONUCLIDE INPUT READ FROM EXTERNAL FILE USER FILE  TempSrc1.txt
    NUCLIDE           HALF LIFE       CURIE
  932370  Np237    2.144E+06  yr    9.050E-06
  922380   U238    4.468E+09  yr    1.610E-04
  922360   U236    2.342E+07  yr    2.170E-04
  922350   U235    7.038E+08  yr    2.520E-04
  942400  Pu240    6.561E+03  yr    3.690E-04
  551370  Cs137    3.008E+01  yr    1.170E-03
```

```
  380900  Sr 90    2.879E+01  yr    5.180E-03
  942390  Pu239    2.411E+04  yr    3.840E-03
  922340   U234    2.455E+05  yr    6.710E-03
  250540  Mn 54    3.120E+02  d     5.810E-05
  270600  Co 60    5.275E+00  yr    1.200E-04
  430990  Tc 99    2.111E+05  yr    2.850E-06
  511250  Sb125    2.759E+00  yr    2.990E-04
  551340  Cs134    2.065E+00  yr    1.190E-04
  551350  Cs135    2.300E+06  yr    2.020E-06
  581440  Ce144    2.849E+02  d     5.980E-04
  631540  Eu154    8.601E+00  yr    1.800E-04
  631550  Eu155    4.753E+00  yr    3.030E-04
  952410  Am241    4.326E+02  yr    6.010E-04
  922320   U232    6.890E+01  yr    6.680E-05
  922330   U233    1.592E+05  yr    1.840E-06
  922370   U237    6.750E+00  d     1.080E-05
```

Meteorological Data

```
        MEAN WIND SPEED = 1.040E+00 (m/s)   STACK HEIGHT = 0.000E+00 (m)
    MIXING LAYER HEIGHT = 4.000E+02 (m)      AIR DENSITY = 1.099E+03 (g/cu m)
    WET DEPOSITION SCAVENGING COEFFICIENT = 0.000E+00 (1/s)
    DRY DEPOSITION VELOCITIES (m/s)
         SOLIDS = 1.000E-03    HALOGENS = 1.000E-02   NOBLE GASES = 0.000E+00
         CESIUM = 1.000E-03   RUTHENIUM = 1.000E-03
    THERE IS 1 SET OF LEAKAGE CONSTANTS (K1,K2)
          1.000E+00       0.000E+00
    PLUME MEANDER FACTOR = 1.00E+00
    PASQUILL CLASS F METEOROLOGY, MARKEE SIGMA VALUES
    NO BUILDING WAKE CORRECTION MADE
      DOWNWIND DISTANCE     STACK        SIGY        SIGZ        CHI/Q
                          HEIGHT (m)     (m)          (m)        (s/m^3)
         5.000E+03         0.000E+00   6.756E+02   1.099E+01   4.120E-05
    PLUME DEPLETION BY FALLOUT IS INCLUDED
      FRACTION OF PLUME REMAINING AIRBORNE FOLLOWING DEPLETION BY DEPOSITION
        DOWNWIND DISTANCE     SOLIDS   HALOGENS    CESIUM    RUTHENIUM
            5.000E+03       9.104E-01  3.912E-01  9.104E-01  9.104E-01
```

Inhalation Dose Calculation

```
         USING DOSE CONVERSION FACTORS FROM ICRP-72 FOR MEMBERS OF THE PUBLIC
    RESPIRABLE FRACTION =  1.000E+00
    BREATHING RATE =  3.330E-04 (m^3/s)
    RELEASE TIME FOR EXPONENTIAL DECAY FUNCTION =  1.000E+00 (s)
    INTERNAL EXPOSURE TIME PERIOD  =  5.000E+01 (yr)
    LUNG ABSORPTION TYPES SELECTED TO GIVE MAXIMUM DOSE
    INHALATION DOSE CALCULATIONS FOR ADULT AGE
    ICRP-72 INHALATION DOSE (rem)         CHI/Q =  4.120E-05 (s/m^3)
      DOWNWIND DISTANCE = 5.00E+03 (m)     PLUME TRAVEL TIME = 4.81E+03 (s)
           E_50 DOSE                   EQUIVALENT   EFFECTIVE
                         TOTAL Mn                      4.03E-09
                         TOTAL Co                      1.72E-07
```

```
                 TOTAL Sr                    3.83E-05
                 TOTAL  Y                    5.16E-09
                 TOTAL Tc                    1.71E-09
                 TOTAL Sb                    1.66E-07
                 TOTAL Te                    1.39E-11
                 TOTAL Cs                    2.22E-06
                 TOTAL Ce                    1.46E-06
                 TOTAL Pr                    4.77E-10
                 TOTAL Eu                    5.38E-07
                 TOTAL Pb                    1.28E-15
                 TOTAL Bi                    1.60E-16
                 TOTAL Ra                    3.28E-13
                 TOTAL Ac                    3.31E-12
                 TOTAL Th                    7.06E-09
                 TOTAL Pa                    2.35E-12
                 TOTAL  U                    3.28E-03
                 TOTAL Np                    2.09E-05
                 TOTAL Pu                    2.33E-02
                 TOTAL Am                    2.67E-03
        TOTAL     E_50 DOSE =                2.94E-02
```

Gamma Dose Calculation

```
      EXPOSURE TIME =  1.0000E+00 (S)
     CALCULATIONS MADE USING THE SEMI-INFINITE MODEL
                                                 EXTERNAL EDE
    DOWNWIND DISTANCE =  5.000E+03 (M)            DOSE =  1.15E-08 (REM)
```

Execution Time

```
           3.00E-02 SECONDS
```

Table A-4. RSAC output file for HALEU Public dose (14,000 m).

Radiological Safety Analysis Computer Program (RSAC 7.2.0)

Name: INL **Company:** Idaho National Laboratory **Serial:** OISSE-77YKP-GW43J

Computer: INL426810 **Run Date:** 10/25/2018 **Run Time:** 14:27:00

File: HALEU 100 kg batch molten spill at 14000m.rsac

Input

```
* Double-click to specify a title
# Double-click to add a description for this file
#
#
# 100 kg HALEU batch. Molten spill. ARFxRF = 6.0E-3
2000,1,0
# Original file copied to temporary external file
2002,TempSrc1.txt
# 2002,C:\Users\chrib\Documents\Rsac\Input Files\HALEU 100 kg inventory.txt
2999
# Public at 14000m
5000,0
5001,1.04,0.,400.,1.099E3,0.,1
5002,0.001,0.01,0.,0.001,0.001
5101,14000.
5201,1.,0.
5400,2,0.,0.
5410,2,6,0,0.
5999
# Public Dose
7000,1,0,1,0,2,6
7001,3.33E-04,0.,0,0,1.
7002,26
7999
# cloud gamma
9000,1,0.
10000
```

Direct Radionuclide Input

```
        PREVIOUS INVENTORY CHANGED TO THE FOLLOWING VALUES
     RADIONUCLIDE INPUT READ FROM EXTERNAL FILE USER FILE   TempSrc1.txt
    NUCLIDE           HALF LIFE        CURIE
  932370   Np237    2.144E+06  yr    9.050E-06
  922380    U238    4.468E+09  yr    1.610E-04
  922360    U236    2.342E+07  yr    2.170E-04
```

```
922350   U235    7.038E+08  yr    2.520E-04
942400  Pu240    6.561E+03  yr    3.690E-04
551370  Cs137    3.008E+01  yr    1.170E-03
380900  Sr 90    2.879E+01  yr    5.180E-03
942390  Pu239    2.411E+04  yr    3.840E-03
922340   U234    2.455E+05  yr    6.710E-03
250540  Mn 54    3.120E+02  d     5.810E-05
270600  Co 60    5.275E+00  yr    1.200E-04
430990  Tc 99    2.111E+05  yr    2.850E-06
511250  Sb125    2.759E+00  yr    2.990E-04
551340  Cs134    2.065E+00  yr    1.190E-04
551350  Cs135    2.300E+06  yr    2.020E-06
581440  Ce144    2.849E+02  d     5.980E-04
631540  Eu154    8.601E+00  yr    1.800E-04
631550  Eu155    4.753E+00  yr    3.030E-04
952410  Am241    4.326E+02  yr    6.010E-04
922320   U232    6.890E+01  yr    6.680E-05
922330   U233    1.592E+05  yr    1.840E-06
922370   U237    6.750E+00  d     1.080E-05
```

Meteorological Data

```
    MEAN WIND SPEED = 1.040E+00 (m/s)   STACK HEIGHT = 0.000E+00 (m)
MIXING LAYER HEIGHT = 4.000E+02 (m)      AIR DENSITY = 1.099E+03 (g/cu m)
WET DEPOSITION SCAVENGING COEFFICIENT = 0.000E+00 (1/s)
DRY DEPOSITION VELOCITIES (m/s)
     SOLIDS = 1.000E-03    HALOGENS = 1.000E-02   NOBLE GASES = 0.000E+00
     CESIUM = 1.000E-03   RUTHENIUM = 1.000E-03
THERE IS 1 SET OF LEAKAGE CONSTANTS (K1,K2)
      1.000E+00       0.000E+00
PLUME MEANDER FACTOR = 1.00E+00
PASQUILL CLASS F METEOROLOGY, MARKEE SIGMA VALUES
NO BUILDING WAKE CORRECTION MADE
  DOWNWIND DISTANCE     STACK         SIGY         SIGZ         CHI/Q
                      HEIGHT (m)     (m)          (m)         (s/m^3)
     1.400E+04         0.000E+00   1.498E+03   1.508E+01   1.355E-05
PLUME DEPLETION BY FALLOUT IS INCLUDED
  FRACTION OF PLUME REMAINING AIRBORNE FOLLOWING DEPLETION BY DEPOSITION
    DOWNWIND DISTANCE     SOLIDS   HALOGENS    CESIUM    RUTHENIUM
        1.400E+04       8.587E-01  2.179E-01  8.587E-01  8.587E-01
```

Inhalation Dose Calculation

```
     USING DOSE CONVERSION FACTORS FROM ICRP-72 FOR MEMBERS OF THE PUBLIC
RESPIRABLE FRACTION =  1.000E+00
BREATHING RATE =  3.330E-04 (m^3/s)
RELEASE TIME FOR EXPONENTIAL DECAY FUNCTION =  1.000E+00 (s)
INTERNAL EXPOSURE TIME PERIOD  =  5.000E+01 (yr)
LUNG ABSORPTION TYPES SELECTED TO GIVE MAXIMUM DOSE
INHALATION DOSE CALCULATIONS FOR ADULT AGE
```

```
    ICRP-72 INHALATION DOSE (rem)          CHI/Q =  1.355E-05 (s/m^3)
      DOWNWIND DISTANCE = 1.40E+04 (m)     PLUME TRAVEL TIME = 1.35E+04 (s)
            E_50 DOSE                    EQUIVALENT   EFFECTIVE
                        TOTAL Mn                      1.25E-09
                         TOTAL Co                      5.33E-08
                         TOTAL Sr                      1.19E-05
                         TOTAL  Y                      4.42E-09
                         TOTAL Tc                      5.31E-10
                         TOTAL Sb                      5.14E-08
                         TOTAL Te                      1.21E-11
                         TOTAL Cs                      6.88E-07
                         TOTAL Ce                      4.54E-07
                         TOTAL Pr                      1.54E-10
                         TOTAL Eu                      1.67E-07
                         TOTAL Pb                      6.10E-15
                         TOTAL Bi                      3.61E-16
                         TOTAL Ra                      1.84E-12
                         TOTAL Ac                      6.73E-12
                         TOTAL Th                      6.12E-09
                         TOTAL Pa                      2.08E-12
                         TOTAL  U                      1.02E-03
                         TOTAL Np                      6.49E-06
                         TOTAL Pu                      7.24E-03
                         TOTAL Am                      8.27E-04
                TOTAL     E_50 DOSE =                  9.10E-03
```

Gamma Dose Calculation

```
       EXPOSURE TIME =  1.0000E+00 (S)
      CALCULATIONS MADE USING THE SEMI-INFINITE MODEL
                                                    EXTERNAL EDE
     DOWNWIND DISTANCE =  1.400E+04 (M)              DOSE =  3.57E-09 (REM)
```

Execution Time

```
           2.00E-02 SECONDS
```